UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	:	
	:	Docket No. 50-170
ARMED FORCES RADIOBIOLOGY	:	
RESEARCH INSTITUTE	:	(Renewal of Facility
	:	License No. R-84)
(TRIGA-Type Reactor)	:	

INTERVENOR CNRS's RESPONSE TO

LICENSEE's FIRST SET OF INTERROGATORIES

NOW COMES the Intervenor in the above-captioned case and pursuant to 10 C.F.R. §2.740b, responds to the Licensee's first set of Interrogatories as follows:

INTERROGATORY 1

Answered by Entwisle.

The Intervenor objects to this question. The only relevance it has to this proceeding is whether the Intervenor has legal standing. This issue has already been resolved, and the names and addresses of members were given in the affidavits that were submitted to establish the Intervenor's standing.

INTERROGATORY 2

Answered by Entwisle.

Elizabeth B. Entwisle, Esq.
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INTERROGATORY 3

Answered by Entwisle.

None.

INTERROGATORY 4

Answered by Entwisle.

Entwisle is an attorney and co-author of a study prepared for the President's Council on Environmental Quality, "NRC's Analysis of Nuclear Accidents: Is it Adequate?" March 1980.

Stillman is a physician and physicist who has participated in the Three Mile Island proceedings before the NRC and whose advanced interdisciplinary training in medicine and physics qualifies him to speak about the biological impact of radiation associated with the operation of the Licensee's reactor.

The contentions are based on examination of the Licensee's documents, such as the Draft and Final Audit Reports, and of documents in the Licensee's docket in the Public Documents Room of the NRC, 1717 H Street, N.W., Washington, D.C.

INTERROGATORY 5

Answered by Entwisle.

Same as Interrogatory 4.

INTERROGATORY 6

Answered by Entwisle.

As of the present, the Intervenor has not determined which, if any, expert witnesses will be called to testi., on any contention.

INTERROGATORY 7

Answered by Stillman. Unless designated otherwise, every Interrogatory hereinafter is answered by Stillman.

a. The Licensee describes two DBAs involving clad failures in their Safety Analysis Report (SAR), namely: a "Fuel Element Drop Accident" and "Fuel Element Cladding Failure Accident." In the Drop Accident the fuel element "has been allowed to decay after being taken out of the operating core and placed in storage. The fission products released from the gap will depend upon the temperature of the fuel following two weeks delay. This temperature is expected to be less than 50°C." (See quote in SAR, pp. 6-16.)

For the Cladding Failure Accident they postulate a gap activity of only 1.4 percent (of the total radioactive inventory in the fuel element) and a maximal release of only 0.2 percent of the iodines. Since the temperature needed to volatilize iodine is 183°C (see SAR, pp. 6-16), it follows that this DBA is presumed to occur at a temperature far below 180°C in order to meet their own criteria for maximal iodine release and total gap activity.

b. Throughout the Hazard Summary Report (HSR), peak fuel temperatures above 600°C are never acknowledged. Furthermore, the selected (fuel element) gap activities and potential radioactive gaseous releases are only realistically compatible with much lower temperatures. For example, the very low values for radioiodines that would be released in a cladding failure accident. For the record, the Intervenor believes that during an inadvertent transient the peak fuel temperature could rise several hundred degrees.

c. The Intervenor makes the conservative assumption that the cladding temperature will essentially mirror the fuel temperature, i.e., an adiabatic transfer of heat between them. Since the Adiabatic Model for heat transfer is given in any elementary thermodynamics text it is unnecessary to cite specific references.

d. During an inadvertent transient, the fuel temperature can rise several hundred degrees depending on the exact core conditions (e.g., specific fuel element configuration, position of the control rods, mechanical malfunctions, operator errors, preceding power history, experiments in progress, loss of coolant, placement of the core within the reactor pool).

e. An "inadvertent transient" or power excursion occurs when there is a sudden core insertion of excess reactivity above cold critical to produce a <u>large rapid increase in neutron flux</u>.

1. According to the Licensee's "technical specifications" the maximum step insertion above critical can be as much as 2.8 percent $\Delta k/k$ reactivity in the pulse mode without any potential danger.

2. If the AFRRI-TRIGA reactor is functioning within the permitted specifications, the maximum reactivity transient that could possibly occur (according to the Licensee) would be that produced by the rapid insertion of the entire available amount of reactivity, namely, 3.5 percent $\Delta k/k$ ($5.00) excess reactivity above cold critical (with or without all experiments in place). The maximum power level associated with such a transient is $\leq$ 10,000 MW. The Licensee maintains that based on the operating experience of the Advanced TRIGA Prototype Reactor (ATPR) in the General Atomic laboratories and calculations using the Fuehs-Nordheim mathematical model, "it can be concluded that the rapid insertion of the total excess reactivity of 3.5 percent $\Delta k/k$ would not represent an undue risk."

3. There are several ways in which an "inadvertent transient" could be initiated and trigger a Power Excursion Accident (PEA), such as:

(1) Improper fuel loading - a reactor operator inadvertently inserts a fuel element into the reactor core when it is already critical.

(2) Failure of an experiment - resulting in an instantaneous insertion of excess reactivity (i.e., the radioactivity associated with the experiment itself) to produce a dangerous transient.

(3) A stuck transient rod - if the most reactive control rod (i.e., the transient rod) is stuck out

of the reactor when the core is already loaded to its total excess reactivity.

(4) Pulsing with the transient rod greater than \$3.00 (2.1% $\Delta k/k$) reactivity - after withdrawal of the three standard control rods (previously withdrawn to achieve a steady state power greater than 1 MW).

4. An "inadvertent transient" requires, by definition, an unplanned error or malfunction in the operation of the TRIGA reactor. Such human errors or equipment failures are very often multiplied during the course of any reactor accident. The history of nuclear reactor accidents, in general, is literally replete with examples of a single malfunction or human error compounded by a series of errors and additional malfunctions. Such a set of circumstances could prevent the safeguards that normally control a "planned transient" from functioning properly, i.e., "within the permitted specifications."

f. During a pulse operation that results in a PEA with cladding failures, the Intervenor postulates that both the fuel-moderator matrix and the claddings will have reached temperatures of 900°C or more. In spite of repeated assurances by the Licensee that the built-in and natural safeguards of the AFRRI-TRIGA would prevent fuel temperatures from rising to and above the safety limit, 1000°C, we contend that such safeguards are not foolproof (see our Interrogatory Answers to question 8a, part 3) and further that there must be circumstances under which such temperature elevations are possible. To document this contention the Licensing Board is referred to

1. The AFRRI Hazards Analysis reviewed by the Test and Power Reactor Safety Branch Division of Licensing and Regulation, Docket No. 50-163, p. 3, 1963, which states that if the three standard control rods are withdrawn to obtain a steady state power of 2 MW, then pulsing with the transient rod of \$3.00 (2.1% $\Delta k/k$) reactivity could raise the peak fuel temperature "to about 900°C due to the temperature at the steady state compounded with the temperature increase from pulsing." Clearly, then, pulsing with a transient rod of more than \$3.00 reactivity could raise the peak fuel temperature to well above 1000°C.

2. Calculations have been made to determine the temperature rise in a central TRIGA fuel element if the cooling water is lost instantaneously (see the 1963 GA-2025 Hazards Summary Report for the 250 KW Mark II TRIGA Reactor [Pulsing] located at the Columbia University in New York City). These calculations clearly demonstrate that a LOCA (in this tank-type TRIGA reactor) can result in fuel element temperatures up to 1200°C.

3. The many experiments routinely performed during the last twenty years at the General Atomic Laboratories

in California with TRIGA fuel elements in which temperatures of 1000°C or more were rather easily attained (regardless of negative temperature coefficients).

g. Experiments have been performed on hydrided 10 wt% U-Zr fuel elements that were rapidly heated by induction. "Results indicated that within about 75 sec the surface temperature reached 930° to 970°C with only minor hydrogen evolution. Abruptly thereafter, the surface was observed to crack parallel with the cylindrical axis, with strong outgassing rates" (see "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," M. R. Simnad, pp. 2-18). Another reference is H. H. Hausner and J. T. Schumar ("Nuclear Fuel Elements," p. 84) where surface cracks appeared in fuel element claddings when they were overheated to 900°C or more. In addition to these specific references, a clear general mechanism is present for concluding that cladding failures are "much more likely" at fuel temperatures greater than 400°C, namely, at elevated temperatures there would be a corresponding increase in the total gap pressure (produced by the rise in fission gas pressure, residual air pressure, and the peak equilibrium hydrogen pressure) that would put the cladding under much greater stress.

h. It is not possible to quantitatively assess the risk of a cladding failure at any fuel temperature. If the Licensee knows some exact way of determining such risks (without knowing the actual probability for each component event) then they should share that knowledge with the rest of the world. Temperatures of operating fuel elements well above 400°C may easily be achieved through pulse heating (see R. E. Taylor's "Pulse Heating of Modified Zr-H," U.S. AEC Report NAA-SR-7736, North American Aviation, 1962). Another scenario for fuel temperature elevation is described under the LOCA-induced multiple cladding accident scenario (see Answer to Interrogatory 24, parts d, e, f).

i. Cladding failures are more likely the higher the fuel element temperature. They are less likely at temperatures below 800°C and become much more possible at temperatures of 900°C or more. For references, see Interrogatory answers to both 7.g. (given above) and 24.c. (given below).

j. Repeated activity in the pulse mode may result in pulse beating. If the peak fuel temperature stays below 550°C the emission of radioactive gases is largely controlled by recoil effects which are not very sensitive to the fuel temperature. However, should the pulse beating result in temperatures above 600°C, the process of gas emission into the gap becomes mostly diffusion controlled and results in "greater gap activity." By contrast with the recoil mechanism, diffusional gas emission is extremely temperature sensitive so that gap activity rises rapidly as a function of increasing fuel temperature.

k. "Greater gap activity" implies larger partial pressures of the radioactive gases contained within the gap. If there is no cladding failure and the cladding maintains perfect structural integrity, then "greater gap activity" will not, of itself, result in a greater fission product release. However, any structural disruption resulting in cladding degradation would result in "greater fission product release" if there were "greater gap activity." Thus cracks or penetrations in the cladding would permit the radioactive gases in the gap to stream out under pressure and if the total gap pressures become great enough to approach 1,800 psi, then these excessive pressures would cause additional breaks in the cladding permitting more rapid release of the gap activity. Total gap pressures in excess of 1,800 psi could even cause complete rupture of the cladding without any prior deterioration.

l. Below 400°C the possibility of a cladding failure is relatively independent of the fuel temperature (e.g., the temperature-independent recoil mechanism is operating at temperatures of 400°C or less).

m. 1,200°C. A core history of at least 100 hours of 1 MW steady state operation.

n. A cladding failure is more apt to occur at fuel temperatures above 1,000°C and total gap pressures in excess of 1,800 psi. The greater the temperature and gap pressure, the more likely the cladding failure.

INTERROGATORY 8

a. Accidents that might occur other than those described in the AFRRI-HSR include

1. Fuel element storage rack failures. Because of their very high radiation levels, spent fuel elements are stored under water for shielding purposes. They are therefore stored in aluminum racks under the pool of water in the reactor tank at the AFRRI facility. The Licensee states that "experience shows it requires approximately 67 fuel elements, of the design used at AFRRI, in a close packed array to achieve criticality." The Intervenor would like to know the exact nature of the "experience" that demonstrates the requirement for "approximately 67 fuel elements" to achieve criticality, given the fact that unlike ordinary fuel elements (in most power reactors) that contain only 3% enriched Uranium - 235, the TRIGA fuel elements contain nearly 20% enriched Uranium-235. Furthermore, even by this overly conservative estimate, the AFRRI could conceivably accumulate this number of fuel elements in their reactor pool over the next 20 years if on-site storage (versus Away From Reactor) remains the guiding principle for the handling of high level radioactive waste. There is also the possibility that unforeseen conditions may require the rapid discharge of the full core load of fuel (i.e., 85 fuel elements). What plans has the AFRRI made to handle such an emergency? Would it really be safe to store this number of fuel elements in their aluminum racks inside the reactor tank?

2. Failure of an Experiment resulting in a significant release of radioactive material. This can result from a runaway experiment undergoing activation either within the reactor core (such as the CET) or in the exposure rooms. There have been at least two documented malfunctions that could effect the release of significant amounts of radiation (from a runaway experiment) into the reactor room, namely:

(i) a safety interlock malfunction that occurred on February 1, 1973;

(ii) malfunction of the lead door rotation on July 27, 1976.

Technical Specifications, section I.A.4 states "The reactor room shall be designed to restrict air leakage when the positive sealing dampers are closed." To accomplish such containment "the door to the corridor behind the reactor control room . . . is a double door that is sealed with compressible rubber gaskets and latched. The double doors at the opposite end of the corridor . . . is also sealed with a compressible gasket." Contrary to this specification, as of October 13, 1978, the above doors were not maintained as designed, in that gasket material was missing on both doors preventing fulfillment of the design function (see Notice of Violation, Appendix A, NRC Inspection Report Docket No. 50-170, 10/13/78). Hence any significant radiation release into the reactor room resulting from an experiment failure could have leaked out of the reactor room into the rest of the AFRRI facility. Another example of this breach of reactor room containment occurred on August 26, 1975, due to failure of the Reactor Room ventilation dampers to close when the Continuous Air Monitor (CAM) was alarmed. Both of these examples plainly demonstrate that any radiation released into the reactor room has a distinct possibility of leaking out into the entire AFRRI building and even outside the building into the environment (given a large enough radiation release) thereby endangering the public health and safety (see the AFRRI Abnormal Occurrence Report to the Directorate of Reactor Licensing, dated 9/3/75).

3. Failure of one or more of the "built-in safeguards," such as the:

(i) safety system channels
(ii) safety system settings
(iii) radiation monitoring systems
(iv) "negative temperature coefficient of reactivity" mechanism for automatic shutdown.

The Licensee describes several built-in safeguards (listed above) that would either alert the reactor operator or some automatic mechanism to effect the necessary corrective measures (e.g., initiate a reactor SCRAM, engage the appropriate interlock system, add water coolant, close the ventilation system, etc.) should an accident even threaten. However, the Intervenor contends that human errors coupled with equipment failures can render these safeguards ineffective, as has occurred repeatedly in nuclear reactor accidents nationwide, such as:

(i) the jumpered safety interlocks of Vermont Yankee;

(ii) the Dresden 2 blowdown in 1971;

(iii) the Millstone seawater intrusion in 1972;

(iv) the Brown's Ferry fire in 1975;

(v) the inversion of control rods and the Rancho Seco control rod drive failure in 1975;

(vi) the relief valve that malfunctioned and stuck open during the accident sequence at Three-Mile-Island in 1979.

To demonstrate that such failures (both human and mechanical) can also occur with the TRIGA reactor at the AFRRI, we shall cite several instances of relevant malfunctions involving these safeguards (reported by the AFRRI to the appropriate federal regulatory agency), including:

(a) On February 1, 1973 the Reactor Core Position Safety Interlock System that coordinates the lead door/core movement (to bring the door into near contact with the core shroud) malfunctioned due to a faulty de-energizing relay.

(b) On January 29, 1974 the Fuel Temperature-Automatic Scram System malfunctioned "due to the build-up of high resistance material on the mechanical contacts of the T2 output meter that initiated the automatic scram through a relay."

(c) On August 26, 1975 the Radiation Monitoring System malfunctioned, i.e., the reactor room ventilation dampers failed to close when the Continuous Air Monitor sensing device was manually triggered. "Inspection revealed that two wires in the control box were loose" and that this was the apparent cause of the malfunction.

(d) On July 10, 1979 there was a malfunction of the Pool Water Level Sensing Float Switch that monitors the reactor pool water level in case of an impending LOCA. "The malfunction was caused by wear on the jacketing around the wires leading to the switch which provided a path to ground, thereby circumventing the switch function."

(e) On July 30, 1979 there was a malfunction of the fuel temperature indicators (i.e., fuel element temperature sensing circuit) ostensibly caused by a "floating signal ground with respect to the system ground." The Licensee admits that "since this system monitors the principal safety parameter of the reactor, it was felt that a more secure ground was required."

(f) On August 9, 1979 the reactor exhaust system malfunctioned due to an electrical fire (in the EF-1 cubicle of the motor control center) caused by a power surge due to a faulty transformer.

(g) On March 15, 1980 there was a malfunction of Safety Channel One such that most of the scram indicators on the reactor control console were illuminated. Further, an inspection on March 17, 1980 "revealed that Safety Channel One would not initiate a scram in accordance with the Technical Specifications of Reactor License R-84. "The cause of the malfunction" was attributed to a damaged operational amplifier on a Safety Channel One circuit board "when electrical power had been reapplied to the console after a power outage."

The Licensee alleges that even if there is a power excursion in the AFRRI-TRIGA and the built-in safeguards malfunction, the reactor will automatically shut down due to the prompt negative temperature coefficient (i.e., -0.126% $\Delta k/k$ decrease per 1°C rise in fuel temperature). This automatic shutdown is entirely dependent on the relative amount and energetic state of the hydrogen nuclei within the U-ZrH_x alloy. Thus, any significant deviation of the hydrogen parameters from their expected values or curves will cause a drastic change in either the prompt and steady-state negative temperature coefficients. Such deviation of the hydrogen parameters are likely under accident conditions where large internal pressures and elevated temperatures may produce phase changes within the U-ZrH_x alloy (see a discussion of the "hydride phases" in "The U-ZrH_x Alloy" by M. T. Simnad, February 1980). Such phase changes will affect the vibration frequency, , of the hydrogen nuclei and thereby seriously alter the negative temperature coefficients of reactivity which depend on the transfer of energy quanta (of magnitude h) from warm or excited hydrogen nuclei to slow or thermal neutrons (via elastic collisions).

4. Multiple cladding failure accidents.

Such accidents may result from any one or more of the following:

(i) Defects in the material integrity of the fuel elements themselves.
(ii) Uncontrolled power excursion (or inadvertent transient) in the operating reactor core (PEA).
(iii) Sabotage or a natural accident (e.g., "act of God") involving the AFRRI-TRIGA Reactor.

The Intervenor contends that cladding failures may occur during operation of the TRIGA reactor secondary to inherent defects or weaknesses in the material integrity of the fuel elements themselves. These may go unnoticed in the required annual fuel element inspections by the Licensee. Note, the frequency of these inspections (for the AFRRI) was decreased from six to every twelve months (in May 1972). Whereas three cladding failures have already been reported by General Atomic in their Torrey Pines TRIGA reactor (see the AFRRI Safety Analysis Report), we contend that there is little reason to believe that they cannot happen in Bethesda, Maryland. In fact, there apparently have been at least two reported cladding failures at the AFRRI itself. The first was on August 17, 1964 whereupon a telegram was dispatched to the AEC in which "the institute has determined that at least one TRIGA type pulsing fuel element exhibits cladding failure." The second occurred on October 19, 1967 in which "small amounts of gas bubbles were observed to be released from a C-ring fuel element." The report (to the AEC on October 24, 1967) goes on to say that "the leaking fuel element was taken out . . . and transferred to the fuel element storage rack in the pool." Clearly there have been and will continue to be cladding failures within the core of the operating AFRRI-TRIGA reactor.

The several ways in which PEAs can be initiated was described above (see answer to Interrogatory 7.e., part 3). The common factor in all of these initiating incidents is that there is a sudden insertion of excess reactivity (within an already critical reactor core) to produce a rapid, large increase in neutron flux (i.e., a prompt power excursion or transient) capable of causing cladding failures at elevated fuel element temperatures. As noted above, if the AFRRI-TRIGA reactor is functioning "within the permitted specifications," the maximum reactivity transient that could occur would be that produced by the rapid insertion of the entire available amount of reactivity, namely 3.5% $\Delta k/k$. Based on the operating experience of General Atomic's ATPR and calculations rooted in the Fuchs-Nordheim mathematical model, the Licensee concluded that "the rapid insertion of the total excess reactivity of 3.5% $\Delta k/k$ would not represent on undue risk." This confidence, however, may be misplaced since the ATPR is certainly not identical with the AFRRI-TRIGA and is under considerably more expert scrutiny and experimental control by the General Atomic scientists. Consider the past history of mechanical malfunctions within the core of the AFRRI-TRIGA reactor itself, including:

(i) cladding-damaged fuel elements on August 17, 1964 and October 19, 1967;

(ii) separation of the transient rod from its connecting rod discovered on July 17, 1973;

(iii) crack detected in the top weld of the transient control rod on May 1, 1974;

(iv) detection of a tilted fuel element within the reactor core, reported January 31, 1975;

(v) misalignment of two fuel assemblies occurred on August 22, 1978.

Consequently, one cannot presume that the AFRRI-TRIGA will always function "within the permitted specifications" and therefore is subject to a serious Power Excursion Accident (PEA) involving one or more cladding failures.

The potential for sabotage or terrorist activity was dramatically pointed out in an April 3, 1979 Draft Audit Report by the Defense Audit Service (DAS) charging that frequent security and safety violations were being committed at the AFRRI. Specifically, the draft audit states that although "NRC's inspections have generally shown that AFRRI's security and safety operations have been satisfactory . . . our review showed that frequent safety and security violations were being committed." Even Admiral Robert Monroe, former Director of the Defense Nuclear Agency (DNA), admitted (see The Washington Star, August 14, 1979) that the possibility for sabotage is real when he said: "If a group of heavily armed, desperate men stormed into the building, there'd be nothing out there to stop them." Clearly, any serious explosion within the reactor room that permitted release of the radioactive inventory into the Facility and beyond, would seriously threaten public health and safety.

As for the possibility of an accident, consider an airplane crash into the AFRRI Facility. There are two major airports (National and Dulles) within 15 miles of the National Naval Medical Center producing extremely heavy air traffic above Bethesda, which is more than five times what is considered safe (from plane crashes) for any nuclear site according to the American National Standards Institute. In addition, a helicopter pad, located on-site at the medical center, is less than one-third of a mile from the nuclear reactor. Other types of accidents are also quite possible. For example, less than a thousand yards from the AFRRI Facility a new Metro subway station and tunnel are being constructed. The Intervenor warns that not only the drilling and dynamite explosions during construction, but future train accidents, might result in conditions that predispose the AFRRI-TRIGA to a LOCA resulting from either rupture of the reactor tank itself, damage to the AFRRI cooling tower, or damage to any part of the pumping system. Such a construction or train accident could also affect one or more of the safeguard systems (listed above) thereby potentiating a dangerous PEA.

5. Maximum Credible Accident (class 9 accident) resulting from an

(i) explosive zirconium-steam (water) interaction (at fuel temperature $\geq$1,000°C) following a PEA-induced multiple cladding failure (without a LOCA), or

(ii) explosive zirconium-oxygen (air) interaction (at fuel temperature $\geq$1,000°C) following a LOCA-induced multiple cladding failure.

The several ways in which a PEA could be initiated and lead to multiple cladding failures were outlined above (see answers to Interrogatory 7.e.3. and 24.c.). As indicated, in those multiple cladding failure accidents due to uncontrolled, prompt, power excursions there is likely to be an associated elevation of the fuel-moderator temperatures (within the damaged fuel elements) to 900°C or more. In particular, those fuel elements reaching temperatures above 1,000°C might produce total gap pressures (≥1,800 psi) capable of rupturing their already damaged stainless steel claddings. Rupture of a fuel element cladding would expose hot Zr H_x (the major component of the fuel element) to the tank water. An NRC report indicates that the rate of a violent zirconium-water (or steam) reaction becomes significant at about 900°C (see NRC memorandum to Roger Mattson from R. O. Meyer, dated April 14, 1979, "Core Damage Assessment for the TMI-2," p. 25). For the strongly exothermic reaction of zirconium with steam approximately 140 k cal per g-mole of zirconium is released at 1,000°C. Each fuel element contains nearly 2 kg of zirconium-hydride, hence a pressure explosion within the ruptured fuel elements would essentially strip these elements and release their entire radioactive inventory into the reactor room. This would also lead to a series of chain-like explosions from additional zirconium-steam interactions as well as other chemical explosions (e.g., from ignition of the hot hydrogen chemically reacting with the oxygen in the reactor room) that would ultimately release hundreds of thousands of curies of mixed fission and activation products into the unprotected atmosphere. Unprotected because the AFRRI-TRIGA reactor (unlike a power nuclear reactor) is not enclosed in any reinforced containment dome. Since there are a few hundred pounds of zirconium-hydride within the core of the TRIGA reactor, which is explosively equivalent to almost half a ton of gunpowder when it reacts with water or steam (at temperatures ≥1,000°C), an explosion within the reactor room would disperse radioactive material over a very densely populated area of many square miles.

Perhaps the most serious credible accident that might befall the AFRRI-TRIGA Reactor would begin with a LOCA. The water coolant could swiftly leave through an open water line, rupture the reactor tank and aluminum tank liner, or be pumped out of the reactor pool. Any of these could be initiated by sabotage, inadvertent accident, mechanical malfunction, or human error either individually or in combination (as was noted in authoritative reports of the infamous Three Mile Island Accident). If the water leaves rapidly (approximately 250 gallons per minute) then the fuel element temperature would rise suddenly (see the Hazards Summary Report, GA-2025, 1963, for the 250 kW Mark III TRIGA Reactor at Columbia University). As noted above (see H. H. Hausner and J. F. Schumar in "Nuclear Fuel Elements," p. 84) a sharp temperature fluctuation of this nature is apt to induce multiple cladding failures. Calculations appearing in the Hazards Summary Report prepared by General Atomic scientists for the Columbia-TRIGA Reactor (a smaller but otherwise similar tank-type of nuclear reactor), show the maximum fuel element temperature resulting from a LOCA might reach 1,200°C. Such peak fuel temperatures (i.e., ≥1,000°C) could produce excessive total gap pressures (i.e., ≥1,800 psi) sufficient to rupture several of the already damaged stainless steel claddings. Under these conditions, rupture of the claddings would expose the $Zr\text{-}H_x$ to air (or oxygen) at temperatures of or above the 1,000°C safety limit. According to Professor Earl A. Gulbransen the chemical reaction between $ZrU_{0.034}\ H_x$ with air is even more violently exothemic than the zirconium-water reaction, releasing more than 260 k cal per g-mole of zirconium at 1,000°C. Furthermore, once started, he claims there is no easy way to stop the explosive reaction. In a certain sense, the explosive mechanism becomes auto-catalytic in that a single explosion would rupture more fuel elements releasing additional zirconium and hydrogen which is then available for further explosive chemical interactions with the oxygen in the air. A series of such core explosions would result in dispersing the radiation inventory of the entire AFRRI nuclear facility over a very large area in and around Montgomery County. The public living within the 5-mile ingestion zone (more than 100,000 people) would, in effect, be showered with such radionuclides as Uranium-235, Strontium-89, Iodine-131, Cerium-144, Cesium-134, Yttrium-91, Krypton-85, Strontium-90, and Cesium-137; all with considerable activities and prolonged half-lives.

b. The Intervenor takes great exception to the Licensee's broad allegation that "accidents ranging from failure of experiments to the largest core damage and fission product release considered possible, result in doses of only a small fraction of 10 CFR part 100 guidelines and are considered negligible with respect to the environment." In fact, all of the accidents described above (in Section 8.a. of these Interrogatories) could violate those guidelines and especially the last two scenarios (involving zirconium explosions) would absolutely result in radioactive releases far in excess of the 10 CFR 100 guidelines. Since the HSR and SAR admit to only minimal population exposure (i.e., "doses of only a small fraction of 10 CFR part 100 guidelines") all of the accidents described by the Intervenor should be considered of "greater severity."

c. In order to quantitatively evaluate the risk of any reactor accident you must know the specific probability for each item in the postulated event-tree as well as the reliability of the subsystems involved. Without an adequate data base, calculation of the probability for each component event is virtually impossible. Similarly, there is a lack of reliability data on many of the essential subsystems. Unfortunately, this type of data is not yet available and even if it were, there are associated theoretical controversies that still plague interested scientists and mathematicians, the infamous Rasmussen Report being a case in point. If only we could quantitatively evaluate the risk of accidents this intervention would probably be unnecessary, for then no one would dare put a nuclear reactor in Bethesda.

d. No one person is really qualified to "properly designate" an accident as a DBA for the AFRRI-TRIGA reactor. It would take a team of qualified experts representing several scientific disciplines including nuclear physics, nuclear engineering, materials science, chemical physics, radiation medicine and health physics. These experts should all be Ph.D.s or M.D.s or both. When dealing with a public danger of this magnitude, it behooves us to use the very best talent we can muster to evaluate the true DBAs - whether it be for the AFRRI-TRIGA or any other nuclear reactor.

e. The Federal Guidelines as they presently exist. It would be advisable to include both the EPA as well as the NRC guidelines.

f. The two maximum credible accidents described in Interrogatory Answer 8.a.(5) above should be designated as DBAs for the AFRRI-TRIGA, because they are the two possible "worst case" accidents. The accidents described in the HSR and SAR are also possible but they are almost trivial compared with the magnitude of the two potential explosive zirconium accidents. For documentation, refer to the testimony presented by Professor Daniel M. Pisello at the Environmental Protection Committee of the New York City Council Hearing on the Hazards of Nuclear Power Plants, June 15, 1979.

INTERROGATORY 9

a. It is common knowledge that spent fuel elements from power reactors are stored in racks under water, because they are highly radioactive. If the racks used to store the elements should fail then enough fuel elements may come together at the bottom of the pool to reach criticality (i.e., produce a critical power excursion). It is a problem which currently concerns many nuclear scientists. At the AFRRI the spent fuel elements are stored 12 to a rack within the reactor pool itself and if the elements must be kept at the AFRRI Facility (because of no federally designated or available AFR disposal site) then the total number of spent fuel elements stored in the reactor pool may become a serious hazard. Unlike the fuel elements in power reactors which are only 3% U-235 enriched, the TRIGA fuel elements are nominally 20% U-235 enriched and are consequently a greater threat. It therefore becomes necessary to determine the maximum number of spent fuel elements that are safe to store assuming the optimum reactive geometrical array if they should come together at the bottom of the reactor pool. The Licensee assures us that if fewer than 67 fuel elements come together, nothing can happen. We simply want to know the "experience" and calculations on which this assurance is based. It also becomes very important should the need suddenly arise to dismantle and temporarily close the entire core of fuel elements, about 85. What provisions have been made for such an emergency?

b. Reasonable assurance could come from two sources, namely:
(i) experiments in which TRIGA fuel elements are placed in the contact configuration under water and the actual reactivity measured with local power range monitors (to measure the local power distribution in the fuel element array);

(ii) criticality excursion calculations for the worst possible geometrical array - that can be evaluated by non-government and non-industry scientists.

c. No specific regulation. However, this is a shortcoming that must be corrected immediately since inadequate fuel element storage now looms large as a terribly significant problem for the entire nuclear industry.

d. The contact configuration represents the optimum reactive geometry, that is, the geometric array most likely to achieve a critical power excursion (i.e., criticality).

e. The values obtained for $k_{eff} \leq 0.746$ and $m/m_{crit.} \leq 0.415$, are a "reasonable assurance" that a 12 element configuration would remain subcritical.

f. The Intervenor accepts the data represented in Figure 2 as adequately representing the experience of the Los Alamos Scientific Laboratory and the Oak Ridge National Laboratory in their experiments U-235 enriched fuel elements Memorandum for Record (January 19, 1981) submitted by the Licensee.

g. The Intervenor is satisfied by the data presented (in their Memorandum for Record) that failure of a storage rack, fully loaded with twelve TRIGA fuel elements would present no safety hazard to either operational personnel or the general public. However, if additional storage racks are contained in the reactor pool, they should be limited to two or three.

INTERROGATORY 10

a. An experiment fails when it either results in an instantaneous insertion of reactivity into the reactor core (type I), or there is a release of radioactive material from an experiment undergoing activation in the reactor (type II).

b. Either type I or II, but by an entirely different mechanism for each. If the type I failure resulted in a PEA (depending on the level of reactivity already operative within the core) and cladding failures (from overheating), then these sets of circumstances could lead to an escape of the radioactive gases from the damaged fuel elements into the reactor room and pool water. A type II failure results directly in the release of radioactive material that could also escape into the reactor room.

c. the same as described in Part 10.b. above.

d. Initially in the reactor room. However, if there is a breach of containment (as described in answers to Interrogatory 8.a.) then the radioactive gases could reach other areas of the AFRRI Facility depending on the nature of the containment breach.

e. Please refer to the Federal Regulations for the occupational limits on each radionuclide.

f. The same as described in part 10.b. above.

g. See Federal Regulations.

h. Depends on the specific experiment sanctioned by the AFRRI.

i. Make certain that the confinement safeguards are intact and functioning properly by more frequent, competent, and independent third-party inspections.

INTERROGATORY 11

a. If the rubber gaskets are totally removed then gaseous radionuclides can enter the ventilation system through adjacent rooms or even penetrate through these rooms to the entire facility.

INTERROGATORY 12

a.
b.
c.
d. To answer these questions accurately, we would have to have more detailed information concerning the physical layout and operational history of the AFRRI Facility.

INTERROGATORIES 13, 14, 15, 16, 17, 18, 19, 20, 21, 22

These questions all refer to specific malfunctions and violations incurred by the AFRRI during their operation of the TRIGA reactor. It makes no sense for us, as outsiders, to second-guess information which is more readily available to them through their own documents or by their direct observation and measurement. The charges we have made are a matter of public record and are in full agreement with the designated regulations and technical specifications necessary to operate the TRIGA reactor safely. If the Licensee is serious about trying to remedy these situations by including our technical input, we recommend that they consult with us and the Union of Concerned Scientists on some formal basis.

INTERROGATORY 23

a. The moderating effect of the $Zr\ H_x$ is largely mediated by the hydrogen nuclei. Experiments performed at the Brookhaven National Laboratory (on neutron thermalization by chemically bound hydrogen) gave results for $Zr\ H_x$ compatible with a solid lattice of regular tetra hedra of zirconium atoms with the hydrogen atoms occupying sites at the center of each tetrahedron. The hydrogen lattice vibrations could be described by an Einstein model with a characteristic energy $h\nu \approx 0.130$ electron volts, where he is Planck's constant and ν is the hydrogen lattice vibration frequency (see A. W. McReynolds, M. Nelkin, M. N. Rosenbluth, and W. Whittemore, "Neutron Thermalization by Chemically Bound Hydrogen and Carbon," Proceedings of the Second U.N. International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1-13, 1958, Paper UN/P/1540). The moderating effect of the hydrogen nuclei may be achieved by elastic collisions with fast or slow neutrons;

that is, prompt or fast neutrons can be slowed down or thermalized by giving up a quantum of their energy, $h\nu$, to the sluggish (or cool) hydrogen nuclei, or slow neutrons may be speeded up by receiving the quantum of energy, $h\nu$, from the energetic (or warm) hydrogen nuclei. For the most part $Zr\ H_x$ is not effective in thermalizing neutrons (because $h\nu \gg kT$), but it can speed up neutrons already thermalized (by the hydrogen nuclei in the tank water). Clearly, anything that changes the hydrogen lattice vibration frequency, ν, will alter the "moderating effect of the $UZr\ H_x$ fuel." In turn, the vibration frequency depends on the fuel temperature, the equilibrium hydrogen pressure (between the hydrogen in the fuel-moderator and the gap hydrogen pressure), and the zirconium-hydrogen phase relationships. Damage to a fuel element is likely to affect one or more of these parameters and thereby change ν, which controls the moderating effect of the hydrogen nuclei in the $U\text{-}Zr\ H_x$.

b. Under normal operating conditions a reduction in the moderating effect of the $Zr\ H_x$ would not appreciably affect the reactivity characteristics of a thermal reactor. However, if the fuel temperature goes above 600°C the hydrogen nuclei in the $Zr\ H_x$ ordinarily reduce the reactivity (i.e., reduce the number of uranium fissions) by warming up the neutrons so they are no longer easily captured by the U-235 nuclei. If the moderating effect of the $Zr\ H_x$ is reduced (e.g., by loss of the hydrogen through cracks in the fuel element claddings) then the reactivity characteristics will show a positive increase (i.e., increase the number of uranium fissions). In other words, the protective "warm neutron effect" which would ordinarily decrease the positive reactivity (or equivalently, increase the negative reactivity) is no longer available because of the reduction of the moderating effects usually mediated by the hydrogen nuclei in the zirconium-hydride.

c. The question is irrelevant since the AFRRI-TRIGA is strictly a thermal reactor.

d. Mathematically, the negative temperature coefficient is primarily a function of $\exp{-h\nu/kT}$, so that any change in the hydrogen lattice vibration frequency, ν, will necessarily modify this coefficient of reactivity. As discussed above, in part a, ν is a function of the zirconium-hydrogen phase relationships which, in turn, depends on the fuel moderator temperature and the equilibrium hydrogen pressure. Thus, any core condition that significantly changes these parameters within the fuel elements will affect the negative temperature coefficient of reactivity. Fuel element cladding failures that permit the escape of hydrogen, will undoubtedly affect the equilibrium hydrogen pressure which ultimately reduces the availability of hydrogen nuclei directly and may induce a phase transition indirectly (due to the reduced concentration or density of hydrogen). Thus, cladding damaged fuel elements can profoundly change the effectiveness of the ordinarily protective negative temperature coefficient by removing a substantial number of hydrogen nuclei (the direct effect) and by modifying ν through a phase transition (the indirect effect). This is why the Intervenor contends that whereas the mechanism for the negative temperature coefficient may operate well under ideal conditions, it may not work very well in a real accident situation (e.g., when the fuel elements may be bent, scratched, corroded, and inadequately cooled) in which case the moderating effect of the hydrogen nuclei (within the U-Zr H_x) could be seriously impaired. Therefore, we argue that this automatic shutdown mechanism is, like other so-called "failsafe" mechanisms, not absolutely foolproof.

e. In the context of this contention, a damaged fuel element may be functionally defined as a fuel element that either leaks hydrogen or undergoes an unusual change in the magnitude of its hydrogen lattice vibration frequency, ν, as the peak fuel temperature goes beyond 600°C.

f. Design an experiment in which cladding damaged fuel elements (that leak hydrogen) are rapidly warmed up (to temperatures of nearly 1,200°) by pulse heating. Keep all the other variables in their usual condition. Be sure to do this experiment in a safe place, not Bethesda.

g. About 1% reduction in the total core moderating effect.

h. This depends on one's criteria for significance. If 10% reduction is significant, then about ten damaged fuel elements would be required.

i. These calculations will be presented by direct testimony at the hearing itself.

j. Data from experiments such as those described above in part f of this interrogatory.

INTERROGATORY 24

a. They do not believe multiple cladding failure accidents are credible.

b. Not that we know of.

c. An "uncontrolled power excursion" may be defined as a large, rapid increase in neutron flux resulting from an unscheduled insertion of excess reactivity above cold critical. Associated with any power excursion is the abrupt rise in fuel temperature. The combined effects of the sudden temperature elevation and the large rapid increase in neutron flux, stress the fuel elements (involved in the power excursion) in several important ways, including:

(i) thermal migration stresses - which arise when hydrogen migrates from the higher to the lower temperature regions of the fuel element, causing the colder regions to expand and the hotter regions to contract. This results in a "migration stress" which is in the opposite sign (or direction) of the thermal stress. The brittle nature of zirconium hydride makes it susceptible to "thermal stress cracking." [See Meyer, R. D., and J. G. LeBlanc, "Negative Thermal Expansion in UZr H Reactor Fuel," Trans. Am. Nucl. Soc. 13, p. 2, 1970.] Furthermore, if the radial temperature gradient on the fuel element is asymmetric, bowing of the rod will occur.

(ii) anomoulous oscillations (secondary to sustained cycling) of the fuel elements if during this redistribution (of hydrogen) and bowing change the thermal gradients are altered, as would be the case in an operating reactor, the conditions for sustained cycling are obtained (i.e., anomalous oscillations). The oscillatory behavior was probably due to the clustering of fuel elements under a thermal gradient, followed by an abrupt declustering (caused by the rehydriding of the fuel elements) which applies a force in the opposite direction. [See Simnad, M. T., "The U-Zr H_x Alloy: Its Properties and use in TRIGA Fuel," GA-4314, 1980, pp. 2-17.]

(iii) surface cracks in the fuel elements effected by the escape of hydrogen - measurements and calculations have been reported of hydrogen loss from hydrided U-Zr fuel elements which were rapidly heated by induction to temperatures ranging from 900° - 1,000°C. After 75 sec at those temperatures, there was an abrupt crack in the surface of the fuel elements associated with major hydrogen evolution (i.e., strong outgassing rates). [See Leadon, B. M. et al., "Aerospace Nuclear Safety-Measurements and Calculations of Hydrogen Loss from Hydrided U-ZrH Fuel Elements During Transient Heating to Temperatures Near the Melting Point," Trans. Am. Nucl. Soc., 8, 1965, p. 8.]

(iv) excessive gap pressures - at elevated temperatures there is an increase in all three component gap pressures (i.e., residual air pressure, fission gas pressure, and the peak equilibrium hydrogen pressure) to produce a total gap pressure that may approach or even exceed 2,000 psi. Note, at temperatures above 800°C the equilibrium hydrogen pressure is, by far, the major contributor. Gap pressures of such magnitude, must put considerable stress on the fuel elements.

(v) irradiation stress resulting from the high neutron fluence caused by the power excursion (e.g., a peak power level of 8,000 MW will create a neutron fluence of about 1.0×10^{15} nvt) - large neutron fluxes induce structural flaws in the substance and claddings of the fuel elements. That is, bombardment by energetic neutrons will produce solid state defects which can migrate and coalesce to establish significant weaknesses and degradations in the fuel elements.

d. As described above in Interrogatory Answer 8.a., part 5), a LOCA would result in a sudden, large elevation of fuel temperature which is apt to produce multiple cladding failures by all of the stress mechanisms outlined in part c of this interrogatory (except that the neutron flux is generated by pulsing rather than an uncontrolled transient).

e. The conditions necessary "to breech the integrity of the fuel's cladding" are entirely divorced from the past power history of the AFRRI-TRIGA reactor. However, the immediate power history of the reactor is important in that "a recurrent pulsing mode" of operation is a vital contributing factor.

f. If the IMW TRIGA reactor was not capable of the pulsing operation, it is unlikely that cladding failures would result from a LOCA involving that reactor. It is noteworthy, however, that the long operational history (about 20 years) of the AFRRI-pulsing TRIGA reactor is more likely to have a multiple cladding failure accident simply because of its longevity (i.e., greater accumulation of radiation-induced solid state defects within the exposed fuel elements themselves).

INTERROGATORY 25

a, b, c Liz Entwistle will respond.

D. The Licensee must concede that there are at least three essential conditions to prevent serious fuel element cladding failures, namely:

(i) fuel element temperature should never exceed the safety limit (1,000°C);

(ii) there must be no sudden core insertion of excess reactivity (i.e., >3.5% $\Delta k/k$ excess reactivity above cold critical) to produce a very large, rapid increase in neutron flux;

(iii) the fuel elements must contain the proper ratio of hydrogen to zirconium and the appropriate mixture of phases of the U-Zr H_x, so that the hydrogen can remain an effective moderator and gap pressures do not become excessive (≥1,800 psi).

Violation of one or more of these conditions can result in (multiple) cladding failure accidents (see answers to Interrogatory 8). At fuel temperatures of 1,000°C or more, almost 10% of the radioactive gases contained in the cladding-damaged fuel elements (that have redistributed into the gaps) will then diffuse out through the degradations or breaks in the claddings (see

In fact, the equivalent of one pound of radium per defective fuel element (in gaseous form) could leak out of the cladding-damaged TRIGA reactor. Another danger to public health and safety is contamination of the tank or cooling water by the (20% enriched) uranium and stored fission products in the cladding-damaged fuel elements. For example, in the event of even a single cladding failure the water activity could easily reach a level of 1µCi/cc and that level would remain elevated (because of the small decay constant of iodine-131 and iodine-133, among other reasons), which greatly exceeds the MPCs outlined in 10 CFR Appendix B. This would not only be a hazard to persons in the reactor room (occupational exposure) but if, by human error, the contaminated water leaked out into the sewage system of Montgomery County and the District of Columbia, it could cause considerable damage to the people living in these communities. Now contemplate a multiple cladding failure and the reason for our concern becomes quite understandable. The two worst accidents (i.e., maximum credible accidents) involving zirconium explosions would, of course, release the entire radioactive inventory of the reactor into the environment. Given the population density, clustering of hospitals, schools, churches, etc., in the Bethesda area, such an accident would be sheer catastrophe.

INTERROGATORY 26

a. The Licensee claims that "no waterborne radioactive emissions are generated by routine operations." Where then did the radionuclides found by the Washington Suburban Sanitary Commission reports (see WSSC reports for 1980-81) come from?

b. & c. The Licensee is legally responsible for disposing of its generated solid radioactive waste which included:

(1) contaminated animal carcasses, tissues and wastes,
(2) mixed laboratory wastes,
(3) scintillation vials with scintillant,
(4) filters which collect reactor by-products,
(5) spent radioisotopic targets,
(6) worn-out or spent fuel elements.

Solid radioactive waste is disposed of in two different ways, namely:

(i) transferred to waste disposal contractor in steel barrels for shipment to radioactive waste burial grounds,
(ii) transferred to the NNMC Radiological Safety Department in boxes for incineration.

The Intervenor contends that incineration of these solid wastes on an unrestricted area of the NNMC grounds results in the airborne release of radioactive gases and particulates that endanger the public health. For example, the AFRRI on one occasion reported the transfer of 160 boxes of contaminated waste to the NNMC for incineration, noting that the principal isotope was Sodium -24. The estimated total activity of this isotope (i.e., Na-24) was approximately 1.6 mCi. Now the annual licensed quantity equivalent (per 10 CFR Appendix C) is no more than 10 μCi, so that this represents (more than a one-hundredfold excess) a clear violation of that safeguard. Another example of dangerous accumulated quantities of radioactive solid waste occurred in 1973 when shipment records indicate that 80 Ci of tritium (targets) were transferred offsite. Since the annual licensed quantity equivalent (per 10 CFR 20 Appendix C) is only one m Ci, this again represents an infraction of Federal regulation and a potential public hazard (see NRC Inspection Reports covering 1975-76, specifically Report No. 50-170/March 1977).

d. We define a "probable violation" as one in which a violation is apt to have been committed except that one or more of the following applies:

(i) the appropriate data were not recorded which would have established the violation,
(ii) the appropriate data were recorded but the reports have been doctored or kept secret,
(iii) the instrumentation or method of measurement was inadequate to actually obtain the pertinent data demonstrating the violation.

e. Specific regulatory limits that have been violated by the AFRRI include:

(1) The concentration of the gaseous radionuclide Argon-41 (Ar-41) released from the AFRRI stack in

1962, 1963, and 1964 exceeded the MPCs listed in 10 CFR 20 Appendix B.

(2) The yearly environmental monitoring data (obtained from the AFRRI Perimeter Monitoring System) demonstrate that the AFRRI has exceeded the well-known Federal regulation that the average yearly ambient radiation levels be less than the 0.5 Rem per year for unrestricted areas. This was the case for both 1962 and 1963.

(3) The data (see Table 2 below) clearly demonstrate that the AFRRI has consistently exceeded the annual exposure EPA limit of 25 mRem for unrestricted areas surrounding a nuclear reactor (see EPA Resolution No. 40 CFR 190).

(4) Technical specifications require the AFRRI reactor building ventilation to exhaust to a stack having a minimum elevation of 18 feet above the roof level of the highest building in the AFRRI complex. Contrary to the above, a leak through a stack drain line discharged part of the exhaust at ground level outside the building for a period of several months.

(5) During the period January 1, 1970 to July 1, 1971 the "normal exposure rate" was 0.5 mRad/hr, however, there were several unrestricted areas where the exposure rate rose to 1 mRad/hr or more and at least one specific unrestricted area where the dose rate approximated 5 mRad/hr. The maximal permissible annual exposure, by NRC regulations, is 500 mRem per year. Hence, any person who lived or worked in these unrestricted areas where the dose rate was 1-5 mRad/hr, would have received excessive radiation if they had been exposed for merely 500 hours during the entire year, or about 10 hours per week. Since the Licensee has failed to convincingly demonstrate that this could not have occurred, it represents a clear violation of the ALARA principle (the goal embodied in 10 CFR part 50) as well as the Federal regulation (requiring that no one receive more than 500 mRem per year). The locations of maximum ionization chamber readings were partly in residential areas. Note approximately 50-60% of the area within a one-mile radius of the AFRRI stack is, in fact, residential.

(6) According to 10 CFR 20.201(b) the "Licensee shall make or cause to be made such surveys as may be necessary for him to comply with the regulations in this part." Contrary to this directive, gross beta measurements of liquid effluents to assure compliance with 10 CFR 20.303, "Disposal by release into sanitary sewage systems," made for the period January 1976 to January 1977 were inadequate in that the gross beta measurements were made without the use of beta self-absorption correction in the presence of significant amounts of suspended solid material.

(7) The AFRRI on one occasion reported the transfer of 160 boxes of contaminated waste to the NNMC for incineration, noting that the principle isotope was Sodium-24. The estimated total activity of this isotope (i.e., Na-24) was approximately 1.6 mCi. Now the annual licensed quantity equivalent (per 10 CFR 20 Appendix C) is no more than

10μCi, so that this represents (more than a one hundred-fold excess) a clear violation of that safeguard.

(8) Another example of dangerous accumulated quantities of radioactive solid waste occurred in 1973 when shipment records indicate that 80 Ci of tritium (targets) were transferred offsite. Since the annual licensed quantity equivalent (per 10 CFR 20 Appendix C) is only one mCi, this again represents an infraction of Federal regulation and a potential public hazard.

f. Specific radiation monitoring methods that the Intervenor considers inadequate, include:

(i) The statistical uncertainty in the annual perimeter dose per monitoring station is ± 20 mRad at the 95% confidence level. This is totally inadequate and could easily be remedied.

(ii) The environmental film dosimetry method employed at the monitoring stations detects only external gamma radiation. Thus, the population radiation exposure dose due to the inhalation or ingestion of radionuclides is entirely neglected.

(iii) The particulate radioactivity monitor for airborne radioactive effluents (i.e., a pancake-probe G-M counter) is not isokinetic and therefore cannot be used for any quantitative evaluations.

(iv) The dose rates (using ionization chamber type instrumentation or an alternative) are not determined with sufficient frequency either for restricted or unrestricted areas (both on and offsite).

(v) The "Environmental Sampling and Analysis" program has been criticized for calculational omissions, the manner in which the samples were prepared for analysis, and the type of instrumentation used to perform the analyses. For these and other reasons, this program should be administered by private and public scientific agencies outside of the Department of Defense (e.g., Washington Sanitary Sewage Commission, the EPA, the Sierra Club, etc.).

g. The principle of self-regulation when it comes to radiation monitoring is highly suspect. The public would be better served if such monitoring were left to private scientific laboratories under the authority and inspection of local government agencies. This puts responsibility for public safety and protection precisely in the hands of the local people who need that protection. Federal agencies could advise, fund, and help implement the appropriate radiation monitoring methods when "corrective actions" are truly indicated to prevent violations of regulatory limits. As long as the Licensee itself has the primary monitoring responsibility they are likely to be inadequate.

h. We are not presently aware of detailed "specific corrective actions" being undertaken by the AFRRI. If the Licensee wishes to share such information, we would be ready to comment on its "adequacy or inadequacy" to prevent a recurrence.

i. We believe "probable violations" have occurred in the following instances:

(1) Since the statistical uncertainty in the annual perimeter dose per monitoring station (at the 95% confidence level) is ± 20 m Rad, it is likely that the annual population exposure at several unrestricted area stations has exceeded the EPA limit (i.e., 25 m Rem) for just about every year during the past 20 years of the AFRRI Facility operation.

(2) The absence of data due to omission of internal radiation makes it virtually impossible to evaluate the true population exposure to radiation, let alone determine whether the Federal regulatory limits have actually been exceeded.

(3) The only two AFRRI particulate radioactivity monitoring systems (i.e., the pancake-probe G-M counter and the radioactive gas monitor) are not reliable for quantitative particulate radioactive sampling. Hence, one can only obtain crude estimates of the airborne radioactive particulates that have been dispersed into the environment. The true values may, in fact, have exceeded public safety limits.

(4) The maximum permissible annual exposure, by NRC regulations, is 500 mRem per year. It is very likely that any person who lived or worked in an unrestricted area where the dose rate was 1-5 m Rad/hr, would have received radiation in excess of this "maximum permissible annual exposure."

j. The following is a list of sources used to document the contention raised above, that the past and present operation of the AFRRI reactor has resulted in probable violations of 10 CFR part 20:

(1) See the letter from the AEC to the AFRRI dated October 6, 1961 which predicts that, according to their calculations for Argon-41 concentrations in unrestricted areas, AFRRI will probably not be able to meet the MPC release standards for unrestricted areas as stated in 10 CFR 20, Appendix B.

(2) See Environmental Release Report (AFRRI-TRIGA Reactor) covering the period 1 Jan. 1970 to 30 Sep. 1971, issued on December 14, 1971.

(3) See Inspection Report No. 50-170/77-01-03 that discusses (gaseous effluent) airborne particulate evaluation.

(4) See the AFRRI-TRIGA Reactor Environmental Release Report issued on December 14, 1971 by AFRRI and the DNA.

(5) In its Environmental Impact Appraisal the Licensee notes that the highest average unrestricted area exposure rates corresponding to given airborne releases are 4.1 m Rem/hr for Argon-41, 4.3 m Rem/hr for the combination of both Nitrogen-13 and Oxygen-1-, and 0.5 m Rem/hr for Xenon-133. These are high dose rates and since they admittedly extend to residential areas, it is quite possible that people may have received in excess of 500 m Rem (the regulatory limit) in any given year. It is also another illustration of violation of the ALARA principle originally designed to protect the public from excessive exposure.

(6) See the AFRRI Environmental Release Data and Perimeter Monitoring Reports Docket No. 50-170 (e.g., May 27,

1966 report, September 20, 1966 report, and December 14, 1977 report).

(7) See the AFRRI's written response to Mr. Joe Miller's (from Citizens for Nuclear Reactor Safety) question #11.

(8)

Table 2

Year	Average Annual Perimeter Dose (per monitoring station)
1962	242 mRad
1963	231 mRad
1964	89 mRad
1965	55 mRad

k. The following is a list of sources used to document the contention raised above, that the past operation of the AFRRI Reactor has resulted in actual violations of 10 CFR part 20 (also see answer to part e. where the actual violations are listed).

(1) With respect to the excessive release of Argon-41 please see the AFRRI Airborne Release Reports for 1962, 1963, and 1964; and the AEC Inspection Reports for 1962, 1963, and 1964, in Docket No. 50-170.

(2) The yearly environmental monitoring data (obtained from the AFRRI Perimeter Monitoring System reports) demonstrate that the AFRRI has exceeded the annual federal limit, 0.5 Rem, for the average yearly ambient radiation.

Table 1

Year	Maximum Annual Exposure	Specific Perimeter Station
1962	> 500 mRad	2c
1963	> 500 mRad	16A
1964	116 mRad	2A
1965	112 mRad	16A
1970	76 mRad	16A
1978	30 mRad	11A

(3) The data (see Table 2 above) also clearly demonstrate that the AFRRI has consistently exceeded the annual exposure EPA limit of 25 mRem for unrestricted areas surrounding a nuclear reactor (see EPA Resolution No. 40 CRF 190).

(4) Regarding the ground level leak of gaseous effluent, see NRC Inspection Report conducted on January 11, 1979 (i.e., Inspection Report No. 50-170/79-01).

(5) See Violation Notice of Gross Beta Effluent Analysis based on the NRC inspection of January 12-14, 1977. Also see Inspection Report No. 50-170/77-01-02.

(6) With regard to violations concerning solid radioactive waste see NRC Inspection Reports covering 1975-76, specifically Report No. 50-170/March 1977.

1. The answer to this question is essentially contained in the answer provided to Interrogatory 26, parts f, g, and h, given above.

INTERROGATORIES 27, 28, 2 _ 30

See Testimony of Professor Ernest J. Sternglass to be presented at the Hearings.

INTERROGATORY 31

All of the component questions center on the meaning given to the phrase "highly probable." In order to assess the probability that the MPCs (set forth in 10 CFR 20, Appendix B) have been exceeded, we would need considerable information (not made available) regarding the individual radionuclide concentrations, air flow parameters and meteorological data occurring during those several months.

INTERROGATORY 32

a. Our contention that air convection cooling alone would not be sufficient to cool an operating TRIGA reactor core during and immediately following a LOCA, is based on calculations contained in the 1963 GA-2025 Hazards Summary Report for the 250 kW Mark II TRIGA Reactor (Pulsing) located at Columbia University in New York City.

b. Internal gap pressures could rise to 1,800 psi or more depending on the temperature elevation. This pressure is capable of producing breaks or penetrations of the fuel element cladding (see Figures 2-9, "Equilibrium Hydrogen Pressure over $Zr\ H_{1.65}$ versus Temperature" in the 1980 GA-4314 Report by M. T. Simnad).

c. TRIGA reactors have definitely had cladding failures. However, whether any such failure has ever been the result of a LOCA is unclear.

d. The Licensee asserts that the maximum amount of fission products that could be released in the event of a cladding failure of a single average fuel element in the AFRRI-TRIGA core is less than 7 curies during steady state operation and also 7 curies during pulse operation (following steady state operation). These calculations, in turn, are based on the assumption that the fraction of gaseous fission products (i.e., radioisotopes of Iodine, Krypton, and Xenon) released from $U\text{-}Zr\ H_x$ fuel into the gap between the fuel material and the fuel element cladding is only 0.1%. That assumption is valid, however, only if the fuel temperature is below 550°C where emission of radioactive gases is largely controlled by recoil effects. However, in a LOCA-induced cladding failure the temperature will rise way above 550°C, so that the process of gas emission into the gap becomes mostly diffusion controlled and radioactive gases begin to stream out of the cladding-damaged fuel element. In fact, a LOCA is apt to produce temperatures in excess of 1,000°C, which means that nearly <u>10% of the radioactive gases in the fuel element escape into the gap region</u> between the fuel and the cladding. Practically all of these gaseous radioisotopes will find their way out of the fuel element gap into the reactor room atmosphere through the penetrations in the damaged cladding. Thus, if 0.1% fraction of gaseous fission products results in a release of above 7 curies, then a fraction of nearly 10% would result in the release of <u>at least 500 curies</u> from a LOCA-induced cladding failure of a single average fuel element. The presence or absence of any breach of containment within the reactor room, will determine the amount of gaseous radioisotopes (e.g., Iodine, Krypton, and Xenon) that ultimately leak into the outside environment (see Figure 5-1, "Fractional Release of Gaseous Fission Products from TRIGA Fuel," p. 5-3 (in the 1980 GA-4314 Report by M. T. Simnad).

INTERROGATORY 33

a. Pulse heating leading to sudden elevations of temperature sufficient to cause multiple cladding failures, has been described several times throughout the body of these Interrogatory Answers (e.g., see Answers to Interrogatories
In the specific case of a LOCA-induced multiple cladding failure, the timing aspect is crucial. It is crucial because the loss of the water is not only a loss of coolant, for a TRIGA reactor it is also the loss of the primary moderator since it is the hydrogen nuclei within the water which actually thermalize the prompt neutrons (i.e., the fast neutrons leave the fuel elements and enter the tank water where they give up their excess energy to become slow neutrons capable of initiating more U-235 nuclear fission). Thus, if the loss of tank water occurs too rapidly there can be no pulsing or inadvertent power excursions (transients). Realistically, we calculate that water could leave the tank at a maximum rate of about 250 gallons per minute (in the several possible ways described above). At this rate of moderator (water) loss, pulsing would not be prohibited and yet cooling would be seriously impaired, producing exactly the necessary conditions for cladding failures to occur.

b. Temperatures above 900°C are rarely, if ever, reached by a single pulse. Indeed, it would take many pulses to establish fuel temperatures of this magnitude rapidly. Temperature fluctuations required to produce cladding failures in a reactor core could result by repeated activation of the pulsing mode following a period of steady state operation.

c. In the answer to Interrogatory 32, part d, we established a fission product release of about 500 curies from a single cladding failure accident if the fuel temperature goes above 1,000°C. Thus, even a single cladding failure would release gaseous radionuclides (e.g., Argon, Krypton, Xenon, Iodine) beyond the limits imposed by 10 CFR 20. One should also allow for radionuclide leaks into the water coolant, given a serious cladding failure (see Interrogatory Answer for details). A multiple cladding failure accident would result in a horrendous release of fission and activation products.

d. The temperature history during a LOCA associated with one or more transients depends entirely on a number of conditions, including the rate of loss of coolant (water), the number and magnitude of the pulses or transients, the previous events immediately preceding the accident (e.g., duration of steady state operation), condition of the fuel elements, etc. Given these and other boundary conditions, we might be able to calculate the temperature history very approximately!

INTERROGATORY 34

The medical concern over radiation exposure secondary to inhalation of gaseous radioisotopes (such as the noble gases) continues to engender controversy among physicians and health physicists. Internal emissions have been a topic of medical symposia such as the NIH Conference about one year ago. In order to do this subject technical justice will require considerable scientific explanation and justification. We expect this topic to be discussed by several of our witnesses at the time of the full hearings, including Dr. Irving Stillman, Dr. Ernest Sternglass, and Dr. Irwin Brass.

INTERROGATORY 35

a. The release of any radionuclide (or combination thereof) into the environment that would expose the public to a dose rate of 100 mRem/hr or more for at least one hour (i.e., a total exposure of at least 100 mRem), constitutes, in our opinion, a "significant offsite release."

b. Since the AFRRI nuclear reactor has been operational for the past 18 years, one can estimate that most of the 87 fuel elements have a radioactive inventory equivalent to about ten pounds of radium per element. This inventory specifically includes, among others, such radionuclides as Uranium -235, Strontium-89, Iodine-131, Cerium-144, Cesium-134, Yttrium-91, Krypton-85, Strontium-90, and Cesium-137; all with considerable activities and prolonged half-lives. Thus, if the radioactive contents of even one fuel element were dispersed into the environment by a chemical explosion (described above) the fission and activation products released to an unrestricted area would violate the 10 CFR (part 100) guidelines many times over.

c. Any of the chemical explosions described above in the two "worst-case" scenarios (e.g., hydrogen-oxygen, zirconium-steam and zirconium-oxygen) could trigger a series of such explosions. Since the AFRRI-TRIGA has no containment dome, these types of accidents would widely disperse the radioactive inventory originally contained within the reactor core. These offsite releases would not merely be "significant," they would be catastrophic.

d. This has been detailed in the answer to Interrogatory given above. However, the sequence may again be briefly outlined as follows: the power excursions produce fuel-moderator heating in an abrupt or rapid manner. The acute temperature fluctuations and massive, sudden increases in neutron flux, effects one or more cladding failures, causing a loss of the hydrogen (that migrates to and accumulates in the gaps) through breaks in the element claddings. Thus, the thermalizing effect of the hydrogen within the fuel elements is severely compromised.

e. The loss of hydrogen nuclei from the U-Zr H_x necessarily reduces its moderating effects and via changes in hydrogen density can also induce phase changes, both of which modify the prompt negative temperature coefficient. A reduction in the effectiveness of the negative temperature coefficient to protect the TRIGA reactor during an accident, would have serious consequences.

f. The explosive zirconium-steam reaction requires zirconium, water or water vapor, and high temperatures (see

g. To the best of our knowledge, all of these necessary conditions can exist within an overheated AFRRI-TRIGA nuclear core (see

h. The explosive zirconium-air interaction requires zirconium, air or oxygen, and high temperatures (see

i. To the best of our knowledge, all of these necessary conditions can exist within an overheated TRIGA core following a LOCA (see

j. This is merely a matter of semantics. Obviously we believe that the two maximum credible accidents (described above) should be designated as the design basis accidents for the AFRRI-TRIGA, rather than the two relatively trivial accidents presently designated as such.

INTERROGATORY 36

a. (2) During the period January 1, 1970 to July 1, 1971 the "normal exposure rate" was 0.5 mRad/hr, however, there were several unrestricted areas where the exposure rate rose to 1 mRad/hr or more and at one specific unrestricted area where the dose rate approximated 5 mRad/hr. The maximum permissible annual exposure, by NRC regulations, is 500 mRem per year. Hence, any person who lived or worked in these unrestricted areas where the dose rate was 1-5 mRad/hr would have received excessive radiation if they had been exposed for, at most, 500 hours during the entire year, or about 10 hours per week (see the AFRRI-TRIGA Reactor Environmental Release Report issued on December 14, 1971 by AFRRI and the DNA).

(1) In its EIA, the Licensee notes that the highest average unrestricted area exposure rates corresponding to given airborne releases are 4.1 mRem/hr for Argon-41, 4.3 mRem/hr for the combination of both Nitrogen-13 and Oxygen-15, and 0.5 mRem/hr for Xenon-133. These are high dose rates and since they admittedly extend to residential areas, it is quite possible that people may have received in excess of 500 mRem (the regulatory limit) in any given year.

(3) and (4) have already been discussed.

b. and c. Data contained in Table 1 and Table 2 given above.

d. The dose rates alluded to were determined by ionization chamber instruments, not the film badges at the environmental station monitors. The large dose rates were attributed to excessive X-Radiation coming from a large X-Ray machine (called the Maxitron). However, this interpretation was never verified.

e. If it is a typographical error then it was made by the AFRRI or DNA in the Environmental Release Report issued on December 14, 1971.

f. We believe so. No.

g. This information describes the radioactive inventory contained within the AFRRI-Reactor Core. It was used whenever such information was needed to make a specific point.

h. Answer by Liz Entwisle. No CNRS members live within 600 feet of the AFRRI stack. The Intervenor is without knowledge of the address of any residence within 600 feet of said stack.

INTERROGATORY 37

a. Admiral Robert Monroe, The Washington Star, Tuesday, August 14, 1979; "Colonel MacIndoe," Montgomery Journal, by Sandy Golden, Wednesday, June 27, 1979.

b. The Intervenor is unable to answer this question without access to information not in the public record.

c. Same as b.

d. Each of the cited instances demonstrates a break in the first and last layer of security between the controlled access areas and the public.

Respectfully submitted,

Elizabeth B. Entwisle

Elizabeth B. Entwisle
Counsel for Intervenor

AFFIDAVIT OF ELIZABETH B. ENTWISLE

I, Elizabeth B. Entwisle, being duly sworn, do state:

1. That the Response of Intervenor Citizens for Nuclear Reactor Safety, Inc. to the Licensee AFRRI's First Set of Interrogatories were prepared under my direction and supervision.

2. That the responses therein designated "Answered by Entwisle" were answered by me.

3. That the responses designated "Answered by Stillman" were answered by Dr. Irving Stillman.

4. That the responses are true to the best of my knowledge, information, and belief.

X Elizabeth B. Entwisle

Elizabeth B. Entwisle

SUBSCRIBED AND SWORN TO before me this 24th day of December, 1981.

Donna J. Zebe

Notary Public 7-1-82

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In the Matter of	:	
	:	
ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE	:	Docket No. 50-170
	:	(Renewal of Facility License No. R-84)
(TRIGA-Type Reactor)	:	

CERTIFICATE OF SERVICE

I hereby certify that the foregoing Response of CNRS to Licensee's First Set of Interrogatories and Affidavit was served on the following by depositing in the United States Mail, first class, this 29th day of December, 1981.

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